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SUMMARY OF THE CAMPAIGN TO PROCESS FOUR PROTOTYPE HFIR TARGETS IN THE TRANSURANIUM PROCESSING PLANT

W. T. McDuffee and L. J. King

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W. T. McDuffee and L. J. King

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ABSTRACT

Four prototype HFIR targets, which had been irradiated at the Savannah River Plant to 73% burnup of the 9.57 g of ^{242}Pu originally contained in each of them, were processed in the Transuranium Processing Plant (TRU) to recover the transuranium elements. Purified products contained about 9.2 g of ^{242}Pu , 8.8 g of ^{244}Cm , 130 μg of ^{252}Cf and 30 μg of ^{249}Bk . This was the first campaign in TRU and it served to test the chemical flowsheets, processing equipment and operating procedures for recovering transuranium elements from irradiated targets. Processing included dissolving the targets in nitric acid (with and without fluoride catalyst), recovering and purifying unburned plutonium using anion exchange (nitrate medium) and purifying transplutonium elements using the Tramex solvent extraction process, which features a tertiary amine as the extractant. Satisfactory results were obtained for each of these steps, but numerous minor changes were required in flowsheets, equipment and procedures. Two attempts to demonstrate the Berkex process for separating and purifying berkelium were unsatisfactory; solids in the feed plugged equipment.

1. INTRODUCTION

The Transuranium Processing Plant (TRU) and the High Flux Isotope Reactor (HFIR) have been built at Oak Ridge National Laboratory to produce large quantities of the heavy actinide elements as part of the USAEC Heavy Element Production Program. Transuranium elements are formed in the HFIR by successive transmutation, beginning with ^{242}Pu which is incorporated in aluminum-clad targets. The irradiated targets are processed in TRU to recover the heavier transuranium elements for distribution to research workers, and to recover americium-curium and unburned plutonium and fabricate them into targets for recycle to the HFIR.

About 600 g of ^{242}Pu and 500 g of higher actinides, ^{243}Am and ^{244}Cm , were produced for feed materials for the program by irradiation in the Savannah River reactors of two 10 kg batches of ^{239}Pu .

Some of the ^{242}Pu was fabricated into special Savannah River reactor slugs and prototype HFIR targets and irradiated during the Savannah River High Flux Demonstration¹ to provide transuranium elements for TRU until HFIR became operational. Seven prototype targets were fabricated to provide additional fabrication and irradiation experience for the HFIR target element.³ One was used for destructive testing and 6 were irradiated from 14 to 20 months. Two of the six were dissolved in other experiments.

This report deals with the processing of four of these targets. This was the first campaign in the Transuranium Processing Plant.² The solution power densities and alpha and beta-gamma activity levels were as high as any that are anticipated.

In addition to providing transuranium elements for use by research workers, this campaign served to test the chemical flowsheets, processing equipment and operating procedures for recovering transuranium elements from irradiated targets. The processing steps that were demonstrated were: (1) mechanical removal of the aluminum coolant-flow shroud tube; (2) dissolution and removal of the aluminum jacket and aluminum matrix material using a solution of NaOH and NaNO_3 ; (3) dissolution of about 70% of the actinide oxide residue using HNO_3 ; (4) dissolution of the remaining actinide oxide residue using fluoride-catalyzed HNO_3 ; (5) separation, purification and recovery of plutonium using 3 cycles of anion exchange treatment; and (6) purification of the transplutonium elements using the Tramex solvent extraction process, which features a tertiary amine as the extractant. Satisfactory results were obtained for each of these steps after numerous minor changes were made in flowsheets, equipment and procedures.

The transplutonium actinides were further purified and separated into individual element fractions by successive application of precipitation, ion exchange, solvent extraction and chromatographic elution techniques using temporary equipment (glass) installed in one of the processing cells.⁴

This report contains a summary of the results of the processing campaign, a description of the targets, and a description and analysis of the processes and equipment used for chemical processing.

2. TARGET DESCRIPTION

The configuration of the prototype target rods is shown in Fig. 2.1. Details of the fabrication and preirradiation tests have been published³. The 20-in.-long active column contained 11.32 g of PuO_2 , dispersed in aluminum powder in 35 aluminum-clad pellets. The 0.250-in. OD x 0.571-in.-long pellets had been pressed separately and then loaded into the finned aluminum target tube, which was sealed by welding and collapsed onto the pellets by the application of a hydrostatic pressure of 10 tsi. Finally the aluminum coolant-flow shroud tube was slid over the finned target tube and attached to it by being rolled into the fins at one end.

The four prototype targets that were processed had been irradiated for 19 to 20 months at the Savannah River Plant and had been allowed to decay for 5 to 6 months.

3. SUMMARY OF RESULTS

This was the first processing campaign in the Transuranium Processing Plant. The solution power densities and alpha and beta-gamma activity levels were as high as any that are anticipated. Purified products contained about 9.2 g of ^{242}Pu , 8.8 g of ^{244}Cm , 130 μg of ^{252}Cf and 30 μg of ^{249}Bk .

Fluoride-catalyzed nitric acid was effective in dissolving these targets. However, it is possible that the treatment that was used to remove excess fluoride from the dissolver solutions was not adequate. In the Plurix process, excellent recovery of high purity plutonium was achieved by sorbing the plutonium, from nitrate solution, onto Permutit SK anion exchange resin and eluting with dilute HNO_3 . The Tramex process, used to separate actinides from fission product lanthanides, was demonstrated to give good recovery of actinides and adequate decontamination from fission products.

Fig. 2.1. Prototype HFIR Target.

Attempts to separate the berkelium from the other actinides using the Berkex solvent extraction process were not successful. Solids in the feed solutions (possibly some zirconium-based polymer) caused valve plugging problems which prolonged processing, so that much of the berkelium probably reverted to the non-extractable Bk(III).

Typical compositions of process streams are given in Table 3.1. Decontamination factors are given in Table 3.2.

Further purification of the transplutonium elements from the Tramex process and separation of the actinides from each other was done using temporary equipment (glass) installed in cell 5 of TRU; this is reported elsewhere⁴.

Each of the four targets was originally loaded with 11.32 g of PuO_2 (9.57 g of ^{242}Pu). A total of 10.8 g of plutonium (97.6% ^{242}Pu), 13.2 g of curium (93.8% ^{244}Cm) and 4.5 g of ^{243}Am was measured in the dissolver solutions. This indicates a ^{242}Pu burnup of 73%. The isotopic compositions of the plutonium and curium are given in Table 3.3.

About 9.2 g of plutonium was recovered after three cycles of ion-exchange processing. Typical decontamination factors for gross gamma activity were 45 across one cycle of ion-exchange processing and 7.2×10^3 across two cycles. The final purification cycle was completed in an unshielded glove box. The ^{244}Pu content (about one-half percent) was about 5 times as high as in material previously available. It was later processed through the electromagnetic separators at the Y-12 plant to produce about 8 mg of plutonium containing 73% ^{244}Pu and about 3 mg of 31% ^{244}Pu .

The Tramex solvent extraction process was used to separate the transplutonium elements (americium, curium, berkelium and californium) from fission products and ionic impurities. Extraction and stripping losses, which were based on ^{244}Cm analyses, averaged 0.1 and 0.3% of the feed when the system was operating normally. Transplutonium product was decontaminated from ruthenium, zirconium-niobium, and cerium by factors of 60, 10 and 2.1×10^3 , respectively. The principal impurities were lithium, nickel and zirconium which were present in concentrations equal to 7, 0.36 and 0.31 g/g curium, respectively.

Table 3.1. Typical Compositions of Various Process Streams

Process Stream	Actinides			Radionuclides					Ionic Materials						
	²⁴⁴ Cm (g/liter)	²⁴³ Am (g/liter)	²⁴² Pu (g/liter)	Gross γ (cpm/ml)	¹⁰⁶ Ru (dpm/ml)	⁹⁵ Zr-Nb (dpm/ml)	¹⁴⁴ Ce (dpm/ml)	^{110m} Ag (dpm/ml)	HNO ₃ (M)	HCl (M)	Zr (M)	Fe (M)	Li (M)	F (M)	Ni (M)
<u>Feed Preparation</u>															
Jacket waste (acidified)	<0.001	-- ^a	--	--	--	--	--	--	~1						
HNO ₃ dissolver solution	1.13	0.36	0.82	9.2x10 ⁹	2.8x10 ¹⁰	1.5x10 ¹⁰	3.8x10 ¹⁰	1.4x10 ⁹	8.7						
F-HNO ₃ dissolver solution	0.37	--	0.36	--	--	--	--	--	9.0						
<u>Ion Exchange</u>															
Feed	0.43	0.14	0.31	3.6x10 ⁹	1.0x10 ¹⁰	6.0x10 ⁹	2.3x10 ¹⁰	7.4x10 ³	8.0		5x10 ⁻⁴	4x10 ⁻⁴		<0.01	
Wash (cold solution)									8.0						
Elutriant (cold solution)									0.65						
Transplutonium product	0.22	0.07	<0.01	--	--	--	--	--	8.0						
Pu product (first cycle)	<0.01	--	0.51	1.6x10 ⁸	1.1x10 ³	4.7x10 ⁷	7.2x10 ⁷	8.1x10 ^E	~2.5						
Pu product (second cycle)	<0.01	--	0.58	6.7x10 ⁵	2.0x10 ⁵	7.6x10 ⁵	2x10 ⁵	2.5x10 ^E	~2.5						
<u>Tramex</u>															
Feed	0.70	~0.3	--	6.5x10 ⁹	2.5x10 ¹⁰	1.8x10 ⁹	2.3x10 ¹⁰	1.2x10 ⁵		0.16			11.0		
Extractant (cold solution)										0.025			--		
Scrub (cold solution)										0.03			10.8		
Strip (cold solution)										8			--		
Raffinate	<0.001	--	--	3.2x10 ^E	--	--	--	--		~0.1			10.9		
Organic waste	<0.001	--	--	2.8x10 ^E	--	--	--	--					--		
Product	0.39	~0.2	--	4.4x10 ⁴	2.1x10 ⁸	1.1x10 ⁸	6.3x10 ⁶	3.3x10 ⁷			2x10 ⁻³		0.43		0.005

^aDash indicates that concentration was not measured.

Table 3.2 Decontamination Factors During Processing of
the Four HFIR Prototype Targets

System	Gross γ	^{106}Ru	$^{95}\text{Zr-Nb}$	^{144}Ce	$^{110\text{m}}\text{Ag}$	^{244}Cm
<u>Ion Exchange</u>						
Plutonium product						
First cycle	45	125	180	60	15	1500
Second cycle	160	64	83	217	25	33
Two cycles	7.2×10^3	8×10^3	1.5×10^4	1.3×10^4	380	5×10^4
Transplutonium product	~ 1	~ 1	7	~ 2	~ 1	--
<u>Tramex</u>						
Transplutonium product	75	60	10	2.1×10^3	20	--

Table 3.3 Isotopic Compositions of Curium and Plutonium in
the Four HFIR Prototype Targets

<u>Curium</u>		<u>Plutonium</u>	
Isotope	Atom %	Isotope	Atom %
^{244}Cm	93.81	^{238}Pu	0.003
^{245}Cm	0.56	^{239}Pu	0.003
^{246}Cm	5.44	^{240}Pu	1.860
^{247}Cm	0.10	^{241}Pu	0.075
^{248}Cm	0.09	^{242}Pu	97.59
		^{244}Pu	0.472

Mechanical operation of the Tramex solvent extraction system was irregular because of failures of minor equipment items. The most troublesome items were frequent failures of solenoid valves in the drive systems for pumps and pulsers and plugging of air-bleed orifices in instrument systems.

Two attempts were made to recover the berkelium from the Tramex product using the Berkex flowsheet. In this flowsheet, the Tramex product is converted to the nitrate system, KBrO_3 is added to oxidize the berkelium to Bk(IV) and berkelium is extracted into 0.6 M di(2-ethylhexyl) phosphoric acid (HDEHP) in purified decane. Only a minor portion of the berkelium was separated and recovered. The presence of solids in the feeds resulted in equipment plugging which delayed the processing such that the oxidant was probably destroyed by chemical action and/or radiolysis. Subsequently the bulk of the curium was removed by LiCl-based anion exchange and the berkelium was recovered using Berkex.⁴ We think the process would have worked in the two runs if the delays had not occurred. However, much development work remains before the effects of many variables in the process are understood.

4. DESCRIPTION OF PROCESSING

The targets had been irradiated to less than 75% burnup of the original ^{242}Pu . Future targets, in which the burnup will be higher, will be dissolved in HCl. However, preliminary experiments⁵ had indicated that fluoride-catalyzed HNO_3 would be required to obtain complete dissolution at these burnup levels. Because the F- HNO_3 system is extremely corrosive to Zircaloy-2, the dissolution was done in a temporary stainless steel dissolver system and a fluoride-removal step was included in the feed preparation procedure. The subsequent processing step, plutonium recovery, was also conducted in the nitrate system. Fig. 4.1 is a block diagram of the major processing steps that were performed during the campaign.

Fig. 4.1. Block Diagram of Processing Steps Performed on Prototype Targets.

4.1 Target Dissolution

The first target was dissolved completely to demonstrate the techniques. Then, the last three targets were dissolved as a unit. The coolant-flow shroud tubes (outer cans) were cut off to minimize the amount of aluminum to be dissolved and the aluminum jacket and matrix material was dissolved in caustic-nitrate solutions which were decanted through a filter to leave a residue of actinide oxides. Only about 70 percent of the actinide residue dissolved in concentrated HNO_3 . In each case, this was transferred out of the dissolver for further processing. Then fluoride-catalyzed nitric acid (about 0.05 M fluoride) was added to the dissolver to complete the dissolution. The two fluoride-bearing solutions were processed separately to remove the fluoride and then combined for subsequent processing.

Aluminum Removal

A target with the outer can removed was put into the stainless steel dissolver, a solution of 6 M NaOH-3 M NaNO_3 was added to dissolve the aluminum, and the dissolver was heated to initiate the dissolving, which proceeded, exothermally, to completion. After the dissolver and contents had cooled, about 30 minutes was allowed for the actinide oxides, which were not dissolved, to settle. The aluminum-bearing solution was decanted through a sintered-metal filter which was then back-washed into the dissolver. The residue was washed using 2 M NaOH and water. The aluminum removal treatments for the last three targets were done sequentially and the actinide oxide residue from all three targets was accumulated in the dissolver. Less than 0.1% of the actinides was found in the filtered aluminum solutions.

Actinide Residue Dissolution

The residue of actinide oxides was treated by boiling for about 2 hours in 15.8 M HNO_3 . About 70 percent of actinides dissolved. The excess acid was removed from the resulting solution by evaporation in the dissolver (in the presence of the undissolved actinide oxides) to about 50% of the original volume. The solution was diluted to the

original volume with 5 M HNO_3 , was transferred out of the dissolver through the sintered metal filter, and was adjusted to feed specifications for the plutonium recovery process.

The actinides which had not been dissolved in fluoride-free acid were dissolved subsequently using nitric acid which contained fluoride as a catalyst. In the case of the one target, the dissolution in fluoride-catalyzed acid immediately followed the fluoride-free dissolution step. In the case of the three-target dissolution, the undissolved actinides were left in 5 M HNO_3 for 26 days prior to dissolution using fluoride-catalyzed acid.

In both cases concentrated nitric acid was added to the dilute acid in the dissolver and the solution was evaporated until the acid concentration was 15.8 M. NaF was added to make the solution about 0.05 molar in fluoride and 15 minutes later $\text{Al}(\text{NO}_3)_3$ (to make the solution 0.05 molar in aluminum) was added to inhibit corrosion of the dissolver. This solution was refluxed for one hour and then concentrated. A solution of boric acid was added to complex the fluoride and solubilize any actinide fluorides present.

After the last dissolution, fluoride-catalyzed nitric acid was refluxed in the dissolver to confirm that the actinides had been completely dissolved. There was no additional dissolution of actinides.

We discovered later that the fluoride catalyst may not have been required to dissolve the residue from the original three-target dissolution. Samples taken before and after the treatment with fluoride catalyst indicated that dissolution had been complete before the catalyst was added. Apparently, standing in 5 M HNO_3 for 26 days followed by concentration of the acid to 15.8 M had been sufficient.

Fluoride Removal

The fluoride-catalyzed dissolver solutions were evaporated to the minimum practical volume (approx. 2-3 liters depending on the dissolved salt content) and pumped into a hydroxide solution which was being air sparged. The resultant slurry was digested for 3 hours at 100°C. Sodium hydroxide was used in precipitating the actinides from the solution from

the single target; however, this caused excessive sodium content in the product and LiOH was used for the solution from the 3-target dissolution. Subsequent laboratory tests indicated that fluoride is not as effectively removed by this precipitation process when LiOH is used in place of NaOH. The alkaline slurry from the single target was stored for 6 weeks prior to the fluoride removal step; the slurry from the 3 targets was processed immediately after the precipitation. In both instances the washed solids after filtration were dissolved in 10 M HNO_3 . The resulting solutions were combined and the actinides were again precipitated using 5 M LiOH. This slurry was then digested at 100°C for 3 hours and was filtered and washed as before. The final crude actinide cake was dissolved in 10 M HNO_3 and the solution was transferred to an evaporator for conversion into feed for the plutonium ion-exchange system. At this time, we had no method for analysis of the fluoride content of radioactive solutions, so the fluoride content was unknown.

4.2 Plutonium Ion Exchange (Plurix)

Plutonium was separated from the transplutonium actinides, and also decontaminated, using a nitrate-based anion exchange process reported by Ryan and Wheelwright.⁶ Throughout the actual processing which is described below, the temperature of the resin was kept at 50 to 60°C.

The plutonium products from two first-cycle runs were combined and processed through a second ion exchange cycle. The equipment was flushed extensively between cycles and new resin was used for the second cycle. The flowsheet was the same for both cycles.

Feed Adjustment

The solution of the actinides was adjusted to about 8 M HNO_3 by evaporation and dilution. The plutonium valence was adjusted to Pu(IV) by adding ferrous sulfamate to first reduce all plutonium to Pu(III) and then adding NaNO_2 and digesting the mixture at 80°C for 30 minutes to re-oxidize to Pu(IV). The bulk of the nitrous gases formed in the re-action was driven from solution during the digestion step; this minimized

gas evolution in the resin bed during the plutonium sorption step which followed. One gram of stable silver (as AgNO_3) was added to second cycle feed solutions to improve decontamination from $^{110\text{m}}\text{Ag}$.

Plutonium Loading

The adjusted feed solution was metered at from 2 to 3 liters/hr to a 3-liter bed (2.4 in. diam x 37 in. long) of 20 to 40 mesh Permutit SK resin which had been previously converted to the nitrate form and washed using 8 M HNO_3 . This represents an area flow rate of about $1.5 \text{ ml min}^{-1} \text{ cm}^{-2}$. The plutonium was effectively sorbed on the resin at HNO_3 concentrations ranging from 6 to 8 M.

Washing

Thirty liters of 8 M HNO_3 was metered through the column at 5 liters/hr ($3 \text{ ml min}^{-1} \text{ cm}^{-2}$) to displace residual feed solution and to wash lightly-sorbed impurities from the loaded resin. Total measured losses of plutonium to the combined loading and washing effluent, which contained the transplutonium elements, were less than 0.5% of the feed. This solution was evaporated to about 4 to 5 liters volume in preparation for adjusting it to feed specifications for the Tramex solvent extraction process.

Elution

Purified plutonium was eluted from the resin bed using 12 liters of 0.65 M HNO_3 metered at 2 to 3 liters/hr ($1.5 \text{ ml min}^{-1} \text{ cm}^{-2}$). From 95 to 105% of the plutonium that was measured in the feed was found in the eluted product; such variations reflect the difficulty in analyzing the feed solutions for plutonium and necessitated second elutions to confirm that plutonium recovery was complete.

Further Purification of Plutonium

Following two cycles of purification in the cells, the plutonium was sufficiently free of fission product impurities to permit further

processing through a third ion exchange cycle in an unshielded glove box in new, clean, laboratory glass equipment. This was a repetition of the second cycle except for the plutonium valence adjustment step. Nitrous oxide gas (NO) was bubbled through the solution of second cycle plutonium product while it was being evaporated to about one liter and then for 3 hr more. This effects the valence adjustment to Pu(IV) without introducing iron and sodium. The acidity of the resulting solution was adjusted to 7.5 M HNO₃ and the solution was then passed through a 0.9-liter bed of anion exchange resin (Permutit SK). The loaded resin was washed with 7.5 M HNO₃ and eluted with 0.65 M HNO₃. The dilute plutonium product was concentrated by evaporation to about 1 liter volume and the plutonium was precipitated using NH₄OH. The Pu(OH)₄ was collected and air-dried on a paper filter and sealed in a glass container for transfer to the calutron-separations area. Losses in the ion-exchange purification and the precipitation steps totalled 0.25% of that fed to the resin bed.

4.3 Tramex Solvent Extraction

Americium, curium, berkelium, and californium were decontaminated from the lanthanide rare-earth elements and other fission products using the Tramex solvent extraction process in which a tertiary amine (Adogen 364-HP) in diethylbenzene (DEB) is used as extractant.

Nitrate-to-Chloride Conversion

The raffinate from the plutonium recovery step was converted from the nitrate form to the chloride form. The nitrate ion must be removed so that its concentration in the feed to the Tramex process is less than 0.05 M. A higher concentration is undesirable because it reduces the separation factor between the actinides and lanthanides.

The method that was used for the first target consisted in precipitating the actinides in LiOH. This was digested for 3 hours at 100°C, and the solids were separated by centrifugation. The cake on the centrifuge bowl was washed using 0.5 M LiOH. Then the washed solids were dissolved from the bowl using 3 M HCl. Several contacts were made to

assure dissolution. Measured losses of curium to the supernate and wash totalled less than 1% of the feed.

This method was also used to recover the actinides from a large volume of recycle solution which contained LiCl.

When the centrifuge became inoperable, another method had to be used. The raffinate solutions from the plutonium-recovery processing of the three-target dissolution and of the fluoride-catalyzed dissolutions were concentrated by evaporating to the minimum practical volume to remove as much HNO_3 as possible. The remaining HNO_3 was removed by repeatedly adding HCl and evaporating the solution. Evaporation of 10 volumes of diluted HCl per volume of original nitrate solution was sufficient to reduce the nitrate concentration to below 0.05 M.

Feed Adjustment

Two slightly different procedures were used to prepare feed for the Tramex process. In the new preferred method, which gives smaller volumes of feed, the final solution was evaporated to the minimum practical volume. Sufficient lithium chloride brine (approx. 13 M LiCl) was then added to give 11 M LiCl in the final volume of feed solution. Evaporation of the resulting mixture was resumed and continued until the boiling temperature reached 137°C, indicating a concentration of 11.8 M lithium.

The solution was cooled to below 70°C and acidified to 1 M HCl with 12 M HCl. The solution was then heated to 120°C and digested at this temperature for about an hour to assure dissolution of any zirconium that might have precipitated during the evaporation and to remove excess acid. Final adjustment to Tramex feed specifications (11 M LiCl, 0.2 to 0.3 M HCl, 0.075 M SnCl_2 and 3% methanol) required only the addition of small volumes containing the required SnCl_2 and free HCl, followed by addition of the methanol.

In the earlier method, the solution containing the LiCl was never heated above 120°C. This assured that the free HCl would never be less than 0.1 M and, therefore, zirconium would not precipitate. This caused a large volume because it was necessary to adjust 8 M LiCl (concentration at 120°C) to 11 M LiCl by adding 13 M LiCl brine. This method was abandoned when the newer method was demonstrated.

The minimum volume of prepared feed is limited to about 8 liters for operating convenience; this gives about 24 hours of column operation on radioactive feed. Other restrictions which determine the feed volume are power density (^{244}Cm concentration) and concentration of impurities (such as sodium and zirconium) which must not be allowed to precipitate.

Tramex Equipment

The contactors used for Tramex processing are three pulsed columns that are 1.4 in. ID and have contacting sections which are 5 ft. long. The pulse plates are 0.02 in. thick, are 0.25 in. apart and contain 0.03 in.-diam holes on a triangular spacing that gives about 5% free area. The extraction and scrub columns are of Zircaloy-2 and the stripping column is of tantalum. Pulsers and pumps are diaphragm pumps which are driven by alternately applying air pressure and vacuum.

The Tramex Process Flowsheet

The flowsheet for the Tramex solvent extraction process is shown in Fig. 4.2. The feed is adjusted, as previously described, and pumped at 0.4 liter/hr to a point near the top of the extraction column. The extractant, which is 0.6 M Adogen 364-HP (a high-purity trialkyl amine) containing 0.03 M free HCl in diethylbenzene diluent, is fed to the bottom of the extraction column at 1.3 liters/hr. The transuranium elements are extracted much more strongly than the rare earths. Small amounts of the rare earths which are extracted or entrained are scrubbed out with 11 M LiCl-0.03 M HCl solution which is fed at 0.6 liter/hr into the top of the scrub column. The scrub column is operated as an extension of the extraction column. The solvent from the top of the extraction column is fed to the bottom of the scrub column and the aqueous from the bottom of the scrub column is fed to the top of the extraction column. The scrubbed organic is fed to the strip column in which the transuranium elements are back-extracted into 8 M HCl which is fed at a rate of 0.5 liter/hr into the top of the column. An organic backwash is introduced into the bottom of the strip column to remove zirconium from the product stream. The backwash is the same composition as the extractant and is

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Fig. 4.2. Flowsheet for the Tramex Process.

fed at 0.5 liter/hr. The rare earths and other contaminants are collected in the extraction-column raffinate or in the waste organic, both of which are discarded eventually.

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